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memorandum

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MENDF6: A 30-Group Neutron Cross-Section Library Based on ENDF/B-VI

The purpose of this memo is to announce the availability of MENDF6, a new multigroup neutron cross-section library that can be used by many X-Division codes. This memo contains only a very brief summary of the contents and characteristics of the library. A much more detailed report on the library is currently under preparation. The interested reader should call me or send an email (on the open) to nucldata@lanl.gov to request a copy of the report. We also intend to archive documentation on CFS under /x6data/doc/mendf6 and on the open and secure XTM WWW pages.

MENDF5 is a random-access 30-group neutron cross-section library that has been heavily used in calculations made with a variety of X-Division codes. The initial version of MENDF5 appeared ~1980; the most recent version was produced in 1987 and is documented in Ref. 1. MENDF5 is based largely on data from the national Evaluated Nuclear Data File, Version V (ENDF/B-V), supplemented substantially by local evaluations from Group T-2 and modestly by evaluations from LLNL. The format of the MENDF5 library is described in Ref. 2.

Version VI of ENDF (ENDF/B-VI) was initially released in 1990, with subsequent revisions in 1992, 1993, and 1995.[3] A variety of improvements have been made for many nuclides, such as a more physical description of secondary particle energy-angle distributions, extended (in energy and detail) resolved resonance data, isotopic (rather than elemental) evaluations, and self-consistent evaluations for standard reactions. Among the materials of interest that have been re-evaluated are D, ^{6}Li, ^{9}Be, ^{16}O, Fe, Pb, ^{235}U, ^{238}U, and ^{239}Pu.

Continuous-energy cross-section libraries based on ENDF/B-VI for our Monte Carlo codes were released in 1994 and are described in Ref. 4. ENDF/B-VI data testing using these continuous-energy libraries is described in Refs. 4-9. How does ENDF/B-VI perform? Overall, ENDF/B-VI is superior to ENDF/B-V, although specific results are very material and application-dependent. For certain problems, ENDF/B-V outperforms ENDF/B-VI. Evaluators continue to work on improvements to some ENDF/B-VI data sets for future releases (for example, the ^{235}U resonance region is not yet represented as accurately as desired).

A library of multigroup cross sections based on ENDF/B-VI is now available to users of X-Division codes. The library is named MENDF6 and may be found on the secure CRAYs as:

/usr/local/xdata/xsec/multgrp/mendf6.

The library is also archived on both open and secure CFS as:

/x6data/mg/sn/mendf6.

Contact me or a member of your favorite code team with questions about utilizing MENDF6 with a specific code.

MENDF6 contains 30-group neutron cross sections for 167 nuclides (the most recent version of MENDF5 had data for 99 nuclides). The list of nuclides found on MENDF6 is given in Table I. Not surprisingly, data for the overwhelming majority of nuclides on MENDF6 are from ENDF/B-VI (146 of the 167 nuclides - identified with ZAIDs ending in .60). In general, ENDF/B-VI evaluations have been taken from the most recent ENDF/B-VI revision for the particular nuclide, up through and including Revision 3. (In fact, beyond Revision 3 with the inclusion of cross sections for the six natural isotopes of Pd based on evaluations completed just this month by P.G.Young of T-2.) The non-ENDF/B-VI data sets on MENDF6 are from T-2, LLNL, and archival LANL evaluations. Processing of ENDF/B-VI evaluations was accomplished using the NJOY[10] and TRANSX[11] codes. The majority of the NJOY processing was performed by R.E.MacFarlane (T-2).

The format of MENDF6 is identical to that of MENDF5. The same 30-group structure is used. The "TD-Division Weight Function" has been used to collapse all cross sections.[12] There is no upscatter or self-shielding. All cross sections are at room temperature. All cross sections are unnormalized. P0 through P4 scattering matrices are provided. Prompt fission nubar data only are used whenever available. The MENDF format allows for only a limited number of reaction cross-section "slots" for each nuclide. The cross section for every reaction given in each evaluation was included in some "slot." The mapping strategy used was the same as used in the preparation of MENDF5 (see Ref. 13 for details). Reaction-product data[14] are included.

Users must be aware that there are certain applications where the use of MENDF6 is totally inappropriate. This is not peculiar to MENDF6, but rather a characteristic of any multigroup library. Difficulties might result from an actual neutron spectrum much different than was used to collapse the cross sections, or from a situation requiring that the cross sections be self-shielded, or from some other reason. For example, the group structure of MENDF6 is simply not designed for problems dominated by thermal neutrons. If you are concerned about the appropriateness of MENDF6 for your application, please contact us.

Over the years, several libraries have been created that are related to MENDF5. For example, there is a version of MENDF5 with energy-dependent fission chis.[15] There is a version of MENDF5 processed into a format appropriate for a production code important to XNH.[16] There is a version of MENDF5 processed into a format appropriate for

MCNP.[17] Finally, there is a companion library to MENDF5 (named MENDF5G[18]) containing photon-production and photon transport cross sections. At this time, no MENDF6-related libraries of this sort have been prepared.

In closing, we will outline our current plans for future activity in this particular area. First, we will produce the highest-priority MENDF6-related files as described in the last paragraph. Next, we plan to create a library similar to MENDF6, but having many more neutron energy groups. Finally, we plan to explore a new format for X-Division multi-group cross-section libraries to allow increased flexibility, accuracy, and robustness. Please let us know if you have any comments on these plans. Final priorities will be governed by decisions made by the Nuclear Data Working Group.

Table I
Materials on MENDF6

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 1001.60    1002.60    1003.60    2003.60    2004.60    3006.60
 3007.60    4007.55    4009.60    5010.60    5011.60    6000.60
 6012.60    7014.60    7015.60    8016.60    8017.60    9019.60
11023.60   12000.60   13027.60   14000.60   15031.60   16000.60
16032.60   17000.60   18000.35   19000.60   20000.60   21045.60
22000.60   23000.60   24000.60   24050.60   24052.60   24053.60
24054.60   25055.60   26000.60   26054.60   26056.60   26057.60
26058.60   27059.60   28000.60   28058.60   28060.60   28061.60
28062.60   28064.60   29000.60   29063.60   29065.60   31000.60
33075.60   36078.60   36080.60   36082.60   36083.60   36084.60
36086.60   39089.60   40000.60   41093.60   42000.60   43099.60
45103.60   45117.90   46102.65   46104.65   46105.65   46106.65
46107.60   46108.65   46110.65   46119.90   47000.55   47107.60
47109.60   48000.65   49000.60   50120.35   50998.99   50999.99
53127.60   53129.60   54000.35   55133.60   55134.60   55135.60
55136.60   55137.60   56138.60   58140.60   58142.60   63000.60
63151.60   63153.60   64000.35   64152.60   64154.60   64155.60
64156.60   64157.60   64158.60   64160.60   67165.60   68166.60
68167.60   69169.55   72000.60   73181.60   73182.60   74000.60
74182.60   74183.60   74184.60   74186.60   75185.60   75187.60
77000.55   78000.35   79197.60   82000.60   82206.60   82207.60
82208.60   83209.60   90230.60   90232.60   91231.60   91233.60
92232.60   92233.60   92234.60   92235.60   92236.60   92237.60
92238.60   92239.35   93237.60   93239.60   94236.60   94237.60
94238.60   94239.60   94240.60   94241.60   94242.60   94243.60
94244.60   95241.60   95242.60   95243.60   96241.60   96242.60
96243.60   96244.60   96245.60   96246.60   96247.60   96248.60
97249.60   98249.60   98250.60   98251.60   98252.60
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